10CFR50.73

Virginia Electric and Power Company
North Anna Power Station
P. O. Box 402
Mineral, Virginia 23117

February 27, 2007

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D. C. 20555-0001

Serial No.: 07-0014
NAPS: MPW
Docket No.: 50-338
License No.: NPF-4

Dear Sirs:

Pursuant to 10CFR50.73, Virginia Electric and Power Company hereby submits the following Licensee Event Report applicable to North Anna Power Station Unit 1.

Report No. 50-338/2007-001-00

This report has been reviewed by the Station Nuclear Safety and Operating Committee and will be forwarded to the Management Safety Review Committee for its review.

Sincerely,

Dan Stoddard

Daniel G. Stoddard
Site Vice President
North Anna Power Station

Enclosure

Commitments contained in this letter: None

cc: United States Nuclear Regulatory Commission
Region II
Sam Nunn Atlanta Federal Center
61 Forsyth Street, SW, Suite 23T85
Atlanta, Georgia 30303-8931

Mr. J. T. Reece
NRC Senior Resident Inspector
North Anna Power Station

IE22

NRC FORM 366 (6-2004) U.S. NUCLEAR REGULATORY COMMISSION	APPROVEO BY OMB NO. 3150-0104 EXPIRES 6-30-2007
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	Estimated burden per response to comply with this mandatory collection request: 50 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records and FOIA/Privacy Service Branch (T-5 F52), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0066), Office of Management and Budget, Washington, DC 20503. If a means used to impose information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
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4. TITLE
Reactor Trip Due To Steam Generator Low Level Coincident With A Steam Flow Feed Flow Mismatch

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NO.	MONTH	DAY	YEAR	FACILITY NAME	DOCUMENT NUMBER 05000
01	03	2007	2007	-- 001 --	00	02	27	2007	FACILITY NAME	DOCUMENT NUMBER 05000

9. OPERATING MODE: 1

10. POWER LEVEL: 100%

11. THIS REPORT IS SUBMITTED PURSUANT TD THE REQUIREMENTS OF 10 CFR §: (Check all that apply)

	20.2201(b)		20.2203(a)(3)(ii)		50.73(a)(2)(ii)(B)		50.73(a)(2)(ix)(A)
	20.2201(d)		20.2203(a)(4)		50.73(a)(2)(iii)		50.73(a)(2)(x)
	20.2203(a)(1)		50.36(c)(1)(i)(A)	X	50.73(a)(2)(iv)(A)		73.71(a)(4)
	20.2203(a)(2)(i)		50.36(c)(1)(ii)(A)		50.73(a)(2)(v)(A)		73.71(a)(5)
	20.2203(a)(2)(ii)		50.36(c)(2)		50.73(a)(2)(v)(B)		OTHER
	20.2203(a)(2)(iii)		50.46(a)(3)(ii)		50.73(a)(2)(v)(C)		
	20.2203(a)(2)(iv)		50.73(a)(2)(i)(A)		50.73(a)(2)(v)(O)		
	20.2203(a)(2)(v)		50.73(a)(2)(i)(B)		50.73(a)(2)(vii)		Specify in Abstract below or in NRC Form 366A
	20.2203(a)(2)(vi)		50.73(a)(2)(i)(C)		50.73(a)(2)(viii)(A)		
	20.2203(a)(3)(i)		50.73(a)(2)(ii)(A)		50.73(a)(2)(viii)(B)		

12. LICENSEE CONTACT FOR THIS LER

FACILITY NAME	TELEPHONE NUMBER (Include Area Code)
M. P. Whalen, Station Licensing	(540) 894-2572

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX
X	SJ	FCV	W120	Y					

14. SUPPLEMENTAL REPORT EXPECTED		15. EXPECTED SUBMISSION DATE	MONTH	OAY	YEAR
YES (If yes, complete 15. EXPECTED SUBMISSION DATE)	X NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)

On January 3, 2007, at 1803 hours with Unit 1 operating at 100 percent power an automatic reactor trip occurred. The initiating signal was the "B" steam generator (SG) low level coincident with a steam flow greater than feed flow mismatch caused by closure of the "B" main feed regulating valve (MFRV). This resulted in a reactor and turbine trip. Closure of the "B" main feed regulating valve was due to a shorted capacitor on the final control card that provides input to the "B" MFRV. At 1932 hours a 4 hour Non-Emergency Report was made to the NRC in accordance with 10 CFR 50.72 (b)(2)(iv)(B). An 8 hour Non-Emergency Report was also made to the NRC in accordance with 10 CFR 50.72 (b)(3)(iv)(A). This event is reportable pursuant to 10 CFR 50.73 (a)(2)(iv)(A) for a condition that resulted in an automatic actuation of any engineered safety feature including the reactor protection system. This event posed no significant safety implications because the Reactor Protection System and Engineered Safety Features Actuation Systems functioned as designed following the reactor trip. Therefore, the health and safety of the public were not affected by this event.

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NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* **(17)**

1.0 DESCRIPTION OF THE EVENT

On January 3, 2007, at 1803 hours with Unit 1 operating at 100 percent power an automatic reactor trip occurred. The initiating signal was the "B" steam generator (SG) (EIIS System AB, Component SG) low level coincident with a steam flow greater than feed flow mismatch caused by closure of the "B" main feed regulating valve (MFRV) (EIIS System SJ, Component FCV). This resulted in a reactor and turbine trip.

Following the reactor trip the Reactor Protection System (RPS) and all Engineered Safety Feature Actuation System (ESFAS) (EIIS System JE) equipment responded as designed including proper operation of AMSAC, and the Auxiliary Feedwater System (AFW) (EIIS System BA). No other major equipment issues were noted.

At 1932 hours a 4 hour Non-Emergency Report was made to the NRC in accordance with 10 CFR 50.72 (b)(2)(iv)(B) for an event causing actuation of the Reactor Protection System when the reactor is critical. An 8 hour Non-Emergency Report was also made to the NRC in accordance with 10 CFR 50.72 (b)(3)(iv)(A) for an event causing actuation of the Auxiliary Feedwater System.

2.0 SIGNIFICANT SAFETY CONSEQUENCES AND IMPLICATIONS

This event posed no significant safety implications because the RPS and ESFAS systems functioned as designed following the reactor trip. Therefore, the health and safety of the public were not affected by this event.

This event is reportable pursuant to 10 CFR 50.73 (a)(2)(iv)(A) for a condition that resulted in an automatic actuation of any engineered safety feature including the reactor protection system.

3.0 CAUSE

Cause of the automatic reactor trip was the "B" SG low level coincident with a steam flow greater than feed flow mismatch. The initiating signal was caused by closure of the "B" MFRV. Closure of the "B" MFRV was the result of a failure of the final control card. The initial failure of the final control card, Westinghouse 7300 Process Control Card type NCB located in C7-331 which corresponds with mark number 01-FW-FCV-1488, was a shorted capacitor (C42). The failed capacitor had been previously identified as a component requiring replacement during refurbishment of NCB cards in November of 2002. This card had been repaired in October of 2002 and placed in stock. As is the practice, these cards are refurbished on-site and then placed in the stock system for use as necessary. The refurbishment program does not require that as improvements to card refurbishment became known that cards in stock and in service should be assessed to determine level of risk and the appropriate action since they were not addressed as a single point vulnerability. There was no administrative or procedural guidance at that time to assist in the decision making process. The root cause of this event is the Organizational and Programmatic Deficiencies that allowed the card to be placed in service in September 2004 without the new upgrades.

The extent of condition applies to those 7300 system PCBs that have been refurbished/repaired and returned to the stock system, and have not been pulled from stock and upgraded, upon knowledge of additional upgrade requirements. The actions recommended by the root cause evaluation will address this condition. The extent of cause applies to PCBs in all systems, which currently do not have a replacement/refurbishment strategy (PM), while in stock.

4.0 IMMEDIATE CORRECTIVE ACTION(S)

Control Room personnel responded to the event in accordance with emergency procedure E-0, Reactor Trip or Safety Injection. Control Room personnel stabilized the plant using ES-0.1 Reactor Trip Recovery. All safety systems responded appropriately. The unit was stabilized at no-load conditions, the Main Feedwater System was placed in service to all three S/Gs and the AFW System secured and returned to normal AUTO/Standby alignment. Subsequently, Control Room personnel transitioned to 1-OP-1.5 in preparation for unit re-start.

5.0 ADDITIONAL CORRECTIVE ACTIONS

The failed circuit board for the "B" main feed regulating valve, 1-FW-FCV-1488, was replaced. The failed card was analyzed by NAPS Card Repair Facility and capacitor (C42) was found shorted in the power supply circuit.
The circuit boards for the Unit 1 "A" and "C" main feed regulating valves were also replaced.

Unit 1 entered Mode 1 at 1635 hours on January 4, 2007. Unit 1 achieved 100 percent power at 1020 hours on January 5, 2007.

6.0 ACTIONS TO PREVENT RECURRENCE

Implement a high priority replacement/refurbishment strategy for 7300 System PCBs in "critical" control loops for PCBs.

Revise Maintenance Department procedure to ensure that PCBs, in the stock system and in service, are evaluated for refurbishment upon receipt of industry/manufactures recommendations of upgraded components.

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NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* **(17)**

7.0 SIMILAR EVENTS

LER N2-06-001-00 dated 11/16/06, documents an automatic reactor trip from "B" steam generator low level coincident with a steam flow greater than feed flow mismatch caused by closure of the "B" MFRV. Closure of the "B" MFRV was the result of a failed isolator card in the SG water level control system for "B" SG. The isolator card failure was a result of one or more transistors in the power supply circuit of the card. The root cause of the transistor failure is age-related degradation.

LER N2-03-001-00 dated 03/31/03, documents an automatic reactor trip from "C" steam generator low level coincident with a steam flow greater than feed flow mismatch caused by closure of the "C" MFRV. Closure of the "C" MFRV was the result of a failed driver card in the SG water level control system for "C" SG. The driver card failed as a result of a blown fuse. The corrective actions from this event focused solely on the driver cards. Fuses were inspected on both units with repairs made to several cards.

8.0 ADDITIONAL INFORMATION

At the time of this event Unit 2 was in Mode 1 operating at 100 percent power and was not affected by this event.

Component information:
Description: Process Control Type NCB
Manufacturer: Westinghouse
Model No.: 2838A30G01
Serial No.: 918378